ATTICUS - Accident Tolerant fuel Test on the Interaction of Coolant with Uranium Silicide

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***ABSTRACT**: Uranium Silicide (U_3Si_2) has a high density and a high thermal conductivity, combined with a low radiation induced swelling rate. Given these properties U_3Si_2 fuel has been used in research reactors successfully for many years and is therefore considered as a potential high burnup and Accident Tolerant Fuel (ATF) for Light Water Reactors. However, out-of-pile corrosion testing of U_3Si_2 at high temperature has shown that the material could be more prone to water interaction than UO_2. The consequences for a pin-hole failure event of a U_3Si_2 fuel pin during reactor operation should therefore be further investigated, before implementation in commercial reactors is feasible.*

The ATTICUS project, which is performed within a collaboration between the Idaho National Laboratory (INL) and the Belgian Nuclear Research Centre (SCK•CEN), aims at investigating the effects of irradiation on the water corrosion resistance of U_3Si_2. Out-of-pile corrosion testing of un-irradiated fuel is currently underway. This testing includes both static autoclave tests in a variety of water chemistry and temperatures. Additionally, pin hole leak tests of unirradiated fuel pellets are planned in a flowing autoclave. These tests will provide the reference point for the water chemistry that is needed during normal operation to limit the extent of fuel-water interaction in case of a leaking fuel pin. It is anticipated that with some addition of H_2 to the water, the amount of interaction is limited. The amount of H_2 added to limit interaction is similar to the water chemistry of typical pressurized water reactor water chemistry conditions. However, radiation effects during reactor operation that could have an effect on the interaction, such as the presence of fission products, are not present in the out-of-pile experiments. Therefore, the irradiation of U_3Si_2 fuel pins is foreseen in the BR-2 reactor of SCK•CEN, which has the goal of underpinning the impact of the presence of fission products and radiation induced microstructural changes.

The BR-2 material test reactor consists of a beryllium matrix composed of 79 hexagonal channels that can either contain nuclear fuel elements, control rods or experimental devices. As a result of the compact core arrangement at the core mid-plane a high thermal and fast flux is available for experimental research. The active core is about 80 cm in height and is cooled by pressurized (12 bar) water at a rate of 10 m/s on the fuel plates at an inlet temperature of about 35 °C . The reactor has a maximum power level of 100 MW, but typically operates at about 50 MW for cycle lengths of 3 to 4 weeks.

The Pressurized Water Capsule (PWC) irradiation device has been used in the BR-2 extensively in the past for fuel pin testing. The PWC is an instrumented capsule that can be used for base irradiation and transient testing of pins up to 1 m long. A schematic overview of the capsule is given below in Fig.1. The capsule is typically pressurized between 40 - 155 bar. Depending on the fuel pin power level and pressure, boiling of the capsule water occurs on the cladding surface and heat is transferred in the radial direction to the capsule inner surface where condensation takes place. The capsule outer wall is cooled by BR-2 primary water. This water flow is regulated by the Calorimetric Device (CD), which allows to evaluate the thermal balance of the capsule, by measuring the water flow and temperatures. In such manner the linear pin power is derived on-line using the measured deposited heat in the water flow, together with reactor physics calculation of local gamma-heating of the structure components. The latter calculation is calibrated with measurements of a dummy rod irradiation performed prior to the fuel irradiation. Any radial outward heat transfer from the cooling flow to other core structures is minimized by an insulating helium screen. Next to the pin power, the cladding temperature and capsule water temperature are measured on-line by several redundant thermo-couples. Neutron flux detectors allow for verifying the axial

power shape in the fuel pin. Sampling of the capsule water is performed also to verify whether fission products are being released from the pin.

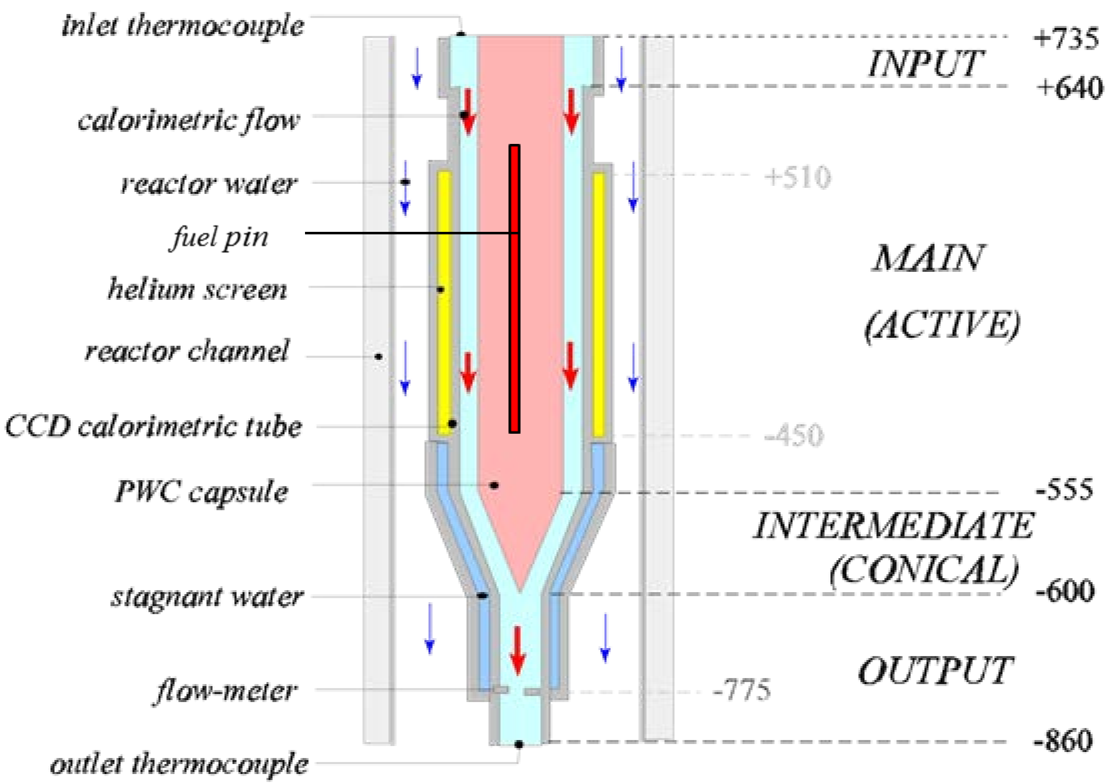

Fig. 1. Schematic overview of the Pressurized Water Capsule and Calorimetric Device to be used for the ATTICUS irradiation.

The fuel test segments will be fabricated by INL. The U_3Si_2 fuel is formulated by arc melting metallic uranium and elemental silicon. Arc melted ingots are crushed and milled into a fine powder, and the powder is cold pressed and sintered to final density. Sintered pellets are machined to final diameter by centerless grinding. The fuel pellets have a diameter of 8.2 mm and a height of 9.8 mm that will be loaded in a 8.4/9.5 mm inner/outer diameter zirconium alloy cladding. In the foreseen irradiation test in the BR-2, the U_3Si_2 fuel test segments will be irradiated in the core to attain a burnup level of 10-40 GWd/t_{HM}. After the burnup accumulation the fuel will be exposed to water at ~300 °C out-of-pile at SCK•CEN. Post experiment examinations are to be performed to quantify the extent of the water-fuel interaction to evaluate the corrosion resistance of irradiated U_3Si_2.

KEYWORDS: ATF, U_3Si_2, irradiation test, BR-2.